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Abstract

IRRADIATION EXPERIENCE WITH ZIRCALOY

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1. Zircaloy-2 is the alloy that was chosen for the pressure tubes and for the fuel cladding in both the NPD and CANDU reactors. Irradiations in the Chalk River research reactors and preliminary examinations of irradiated components from NPD indicate that the alloy's performance accords with the criteria established in the design of both pressure tubes and fuel.

2. Special equipment was designed and developed for measuring pressure tubes in NPD and in NRU loops. It has been shown that diametral creep is negligible and that the movement of fuel bundles causes no significant wear or fretting on the pressure-tube wall.

3. When in use, the pressure tubes are biaxially stressed, and their behaviour has been studied by testing to burst. The burst strength is higher than that calculated from measurements on specimens uniaxially stressed.

4. Fuel clad in Zircaloy-2 has successfully withstood irradiation to 10 000 MWd/tonne, and with hydrogen concentrations in the cladding of more than 200 ppm. Oxidation and hydriding of fuel cladding has sometimes been found to exceed that in equivalent samples not subject to the heat and radiation fluxes. Inferior surface conditions are thought to account, at least in part, for this. The amount of hydrogen absorbed by the sheath is 20 - 40% of the equivalent of the corrosion hydrogen.

5. Zircaloy-2 specimens with hydrogen concentrations up to 250 ppm have been irradiated to 10^{21} n/cm^2. The irradiation increases the tensile strength by 20 - 30% and decreases the elongation by 50%, but the various concentrations of hydrogen have negligible effect on the tensile properties. The impact strength decreases as a function of both neutron irradiation and hydrogen concentration.

6. The creep of cold-worked Zircaloy-2 has been measured in a fast-

neutron flux of 5 to 10 x 10^{12} cm^{-2} s^{-1} (E > 1 MeV) at 14 and 21 kgf/mm^2 and 300°C. In these tests it was observed that the in-reactor primary and early secondary creep was only slightly higher than in the laboratory, but the difference increased with test time. This was because the laboratory creep rates decreased continuously whereas in the reactor a steady rate was reached after approximately 1000 hours.

7. Zirconium hydrides precipitate parallel to the direction in which the grains are deformed and the degree of directionality increases with increasing strain and with heat-treatment temperatures up to 600°C. Annealing at temperatures between 600 and 800°C does not destroy directionality but decreases it. Below a strain of about 7% no directionality is observed.

8. Tensile tests have been carried out on specimens in which the stress was applied perpendicular to hydride platelets. The hydride did not have a marked effect on ductility unless the specimen thickness was decreased to about 10 times the length of the hydride platelets.

9. Corrosion samples have been subjected under irradiation to pressurized lithiated water, to ammoniated steam-water mixtures and to ammoniated superheated steam. Weight gains for these samples were similar to control samples in static autoclaves and not under irradiation. However, the hydriding rates were different and this may be due to different oxygen concentrations.